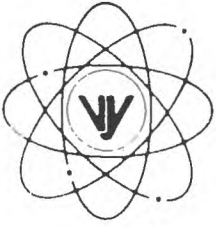


VERMONT YANKEE NUCLEAR POWER CORPORATION



P.O. Box 157, Governor Hunt Road
Vernon, Vermont 05354-0157
(802) 257-7711

July 9, 1998
BVY 98-101

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington D.C. 20555

**Subject: Vermont Yankee Nuclear Power Station
License No. DPR-28 (Docket No. 50-271)
Reportable Occurrence No. LER 98-016, Rev. 0**

As defined by 10CFR50.73, we are reporting the attached Reportable Occurrence as LER 98-016, Rev. 0.

Sincerely,

VERMONT YANKEE NUCLEAR POWER CORPORATION

Michael A. Balduzzi
Plant Manager

cc: USNRC Region I Administrator
USNRC Resident Inspector – VYNPS
USNRC Project Manager – VYNPS
VT Dept. of Public Service

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NRC Form 366 U.S. NUCLEAR REGULATORY COMMISSION (4-95) LICENSEE EVENT REPORT (LER)				APPROVED BY OMB NO. 3150-0104 EXPIRES 04/30/98 ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20566-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.							
FACILITY NAME (1) VERMONT YANKEE NUCLEAR POWER STATION						DOCKET NUMBER () 05000271			PAGE (3) 01 OF 04		
TITLE (4) Reactor Scram on High Water Level as a result of a Stuck Open Feedwater Level Control Valve due to a Cap Screw Lodged Underneath the Valve Disk.											
EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)		
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NO.(S) 05000	
06	09	98	98	-- 016 --	00	07	09	98	N/A		
OPERATING MODE (9)		N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: CHECK ONE OR MORE (11)								
POWER LEVEL (10)		65	20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)		
		20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)			
		20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71			
		20.2203(a)(2)(ii)		20.2203(a)(4)		X	50.73(a)(2)(iv)		OTHER		
		20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		(Specify in Abstract below or in NRC Form 366A)			
.....			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)				
LICENSEE CONTACT FOR THIS LER (12)											
NAME MICHAEL A. BALDUZZI, PLANT MANAGER								TELEPHONE NO. (Include Area Code) 802-257-7711			
COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)											
CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	
NA						NA					
NA						NA					
SUPPLEMENTAL REPORT EXPECTED (14)						EXPECTED SUBMISSION DATE (15)			MO	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)				X	NO

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On 6/9/98 at 0132 hours, during power reduction in preparation for single loop operation in response to a Recirculating Pump motor-generator (MG) high bearing oil temperature, the reactor scrambled due to a turbine trip on high water level. The "A" feedwater level control valve would not close to accommodate the reduced power due to foreign material, a cap screw, being lodged in the valve. The valve could not close beyond approximately 42% from its fully open position. The running Feedwater Pumps tripped on high water level and reactor water level decreased to the low level setpoint such that Primary Containment Isolation System Groups 2, 3 and 5 were actuated. Approximately fifty minutes after the scram, the "C" Feedwater Pump was started but tripped on low indicated minimum flow. Following the trip of the "C" Feedwater Pump, the "A" and "B" Feedwater Pumps attempted to simultaneously auto-start which caused an overcurrent condition on 4KV bus 1 and subsequently tripped the supply breaker to 4KV bus 1 as well as the feeder breaker to 4KV bus 3. This resulted in a loss of 4KV power to the "A" and "B" Feedwater Pumps, the "A" Condensate Pump and the "A" Recirculation Pump plus 480VAC station busses 6, 8, and 11. Due to the loss of power, the "B" Diesel Generator started and assumed the loads on 4KV bus 3 and 480VAC bus 8. When power was restored, bus 6 was cross-tied to bus 7, the turbine bypass valves ramped open which caused a high steam flow scram signal and Group I PCIS isolation. Level control was maintained through the use of the Reactor Core Isolation Cooling system and Torus cooling was in effect. The plant was restarted on 6/13/98 following removal of the cap screw from the feedwater valve and repair of the MG set bearing. The root cause of this event is attributed to foreign material in the feedwater system. All safety systems performed as designed and, therefore, there was no increased threat to public health and safety.

NRC Form 366 U.S. NUCLEAR REGULATORY COMMISSION (4-95)		APPROVED BY OMB NO. 3150-0104 EXPIRES 04/30/98 ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20566-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.			
LICENSEE EVENT REPORT (LER)					
FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REV #	
VERMONT YANKEE NUCLEAR POWER CORPORATION	05000271	98	-- 016 --	00	02 OF 04

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

DESCRIPTION OF EVENT

On 6/9/98 at 0132 hours with the plant at approximately 65% power during power reduction in preparation for single loop operation, the reactor scrammed due to a turbine trip on high water level. A subsequent reactor low level caused the Primary Containment Isolation System (PCIS) (EIS=JM) Groups 2, 3, and 5 to actuate. The power reduction was in response to a Recirculating Pump (EIS=AD) motor-generator (MG) high bearing oil temperature. During the power reduction, the "A" Feedwater Level Control (EIS=SJ) valve would not close sufficiently to accommodate the reduced power because foreign material, a large cap screw, was stuck in the valve and the valve could not close beyond approximately 42% from its fully open position. Reactor level increased and the turbine and the running Feedwater Pumps tripped on high water level. At 0142 hours the scram was reset and the isolations verified.

Approximately fifty minutes after the scram, the "C" Feedwater Pump was started manually but tripped on low flow. Following the trip of the "C" Feedwater Pump, the "A" and "B" Feedwater Pumps attempted to auto-start which caused an overcurrent condition on 4KV bus 1 and subsequently tripped the supply breaker to 4KV bus 1 as well as the feeder breaker to 4KV bus 3 (EIS=EB). This resulted in a loss of 4KV power to the "A" and "B" Feedwater Pumps, the "A" Condensate Pump and the "A" Recirculation Pump MG set plus 480VAC station busses 6, 8, and 11. Due to the loss of power, the "B" Diesel Generator (EIS=EK) started at 0225 hours and assumed the loads on 4KV bus 3 and 480VAC bus 8. 480VAC Bus 6 was cross-tied with 480VAC bus 7 and when this power was restored, the Turbine Auxiliary Oil Pump (EIS=TD) started and the Turbine Bypass Valves (EIS=JI) ramped open causing a high steam flow scram signal and Group I PCIS isolation. A Group I isolation isolates the reactor such that level and pressure control was maintained through the use of the Safety Relief Valves (SRV), Reactor Core Isolation Cooling (RCIC)(EIS=BN) system and the Control Rod Drive system (EIS=AA). The "A" loop of Residual Heat Removal (RHR) system was used to cool the Torus.

The "A" Recirculation Pump MG set tripped when bus 1 was lost and at 0226 hours the "B" Recirculation Pump MG set tripped from a suspected loss of power to the generator tachometer circuit. With both MG sets secured, the reactor vessel experienced stratification which was corrected prior to re-starting the Recirculation Pumps. At 1445 hours the decision was made to de-inert the containment and make an entry to throttle down on the Reactor Water Clean-up (RWCU) system suction valve from the recirculation loop. The RWCU system can take a suction from either a recirculation loop or the vessel bottom drain. This would result in more flow from the vessel bottom drain which would decrease the vessel stratification, or delta temperature, to within the limits for starting a Recirculation Pump. The "A" Recirculation Pump was re-started at 0710 hours on 6/10/98.

During the establishment of level control with RCIC, the SRV's were being used to maintain reactor pressure and at 0302 hours a scram signal was generated due to low reactor water level. Level was subsequently restored to normal using the CRD and RCIC systems. Due to the operation of the SRV's, the Torus volume limit was slightly exceeded and, at 0430 hours, the primary containment was declared inoperable. At 0705 hours, the transfer of torus water to Radwaste was initiated and, at 0725 hours the Primary Containment was declared operable when Torus volume was returned to normal level.

The operating staff initially took manual control of the feedwater level control valve in an attempt to control reactor water level, however, a high water level turbine trip occurred prior to gaining control. Vessel level control was stabilized with the use of RCIC and the CRD systems.

Following plant stabilization from the scram, the plant went to Cold Shutdown while maintenance was performed to correct the feedwater valve and the MG set bearing. Investigation into the source of the cap screw was undertaken but no conclusive source was found.

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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

CAUSE OF EVENT

The root cause of this event is inadequate Foreign Material Exclusion (FME) area control in the Feedwater System.

ANALYSIS OF EVENT

The Reactor Protection System (RPS) provides the required signals to automatically scram the reactor upon receipt of a turbine stop valve closure and functioned correctly to meet its safety design basis.

The RCIC system operates automatically in time to maintain sufficient coolant in the reactor vessel so that the integrity of the radioactive material barrier is not compromised. In this case the RCIC was manually initiated to maintain level control and functioned in accordance with its safety design basis.

The Nuclear System Pressure Relief System prevents overpressurization of the Nuclear System in order to prevent failure of the Nuclear System process barrier due to pressure. The SRV's were used manually several times to control reactor pressure following the scram. These components operated correctly and maintained reactor pressure below all applicable limits.

Protective relaying and overcurrent protection provided the proper bus deenergization of the 4KV busses and 480VAC busses. The diesel generator responded properly by starting and accepting loads from 4KV bus number 3.

If this event had happened at a higher power level it is expected that system and operator responses would have been the same. The higher power level would result in a larger heat load which would require longer times to attain cold shutdown.

The vessel coolant inventory was never reduced to the level requiring actuation of any emergency injection source. Containment isolations which occurred did not challenge the operators' abilities to achieve a stable shutdown condition.

As the systems that responded automatically and those that were actuated manually, all operated in accordance with their design parameters, there was no increased threat to public health and safety.

CORRECTIVE ACTIONS

Immediate

1. The Operators placed the plant in a stable condition using plant Operating and the Emergency Operating Procedures (EOPs).
2. An Event Report was written to ensure that an investigation and root cause analysis was performed.
3. A post trip investigation was conducted by the Operations Department.
4. A task force investigation into the cause, with subsequent corrective action recommendations was performed.
5. Boroscope inspections of both Feedwater Level Control valves and portions of the feedwater piping were conducted. Both valves were disassembled, the cap screw removed from the "A" valve and no further foreign material was discovered. The investigation showed that the cap screw did not come from any system equipment.

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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

6. The "B" Recirculation Pump MG set bearing was replaced.
7. Procedural guidance has been added to station procedures to:
 - a. place the Feedwater Pumps in Pull-to-Lock during transient conditions and only to have one pump in standby when one pump is running or being started.
 - b. provide cautions to be considered when busses need to be cross-tied.
8. An adverse trend Event Report was written by the task force to document the unsatisfactory FME trend.
9. Procedure AP 0048, "Work Planning" was revised to require all Work Orders to have FME controls in place when work involves disassembly of any plant component.

Long Term

1. Review the design of the Reactor Feedwater Pump automatic start logic. The expected completion date is 12/21/98.
2. Develop specific loss of bus procedures. The expected completion date is 3/31/99

ADDITIONAL INFORMATION

Similar events reported to the Commission in the last five years are: LER 95-21, LER 96-19, LER 97-08 and LER 97-23. It should be noted that LER 95-21 reported a plant scram as a result of a turbine trip due to a high water level which resulted from a feedwater regulating valve failure. The failure in this case was an error made during a maintenance activity whereby the valve operator became disconnected from the valve.

CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR: 9807130347 DOC. DATE: 98/07/09 NOTARIZED: NO DOCKET #
FACIL: 50-271 Vermont Yankee Nuclear Power Station, Vermont Yankee 05000271
AUTH. NAME AUTHOR AFFILIATION
BALDUZZI, M.A. Vermont Yankee Nuclear Power Corp.
RECIP. NAME RECIPIENT AFFILIATION

SUBJECT: LER 98-016-00: on 980609, reactor scram on high water level as was noted as result of stuck open FW control valve. Caused by cap screw lodged underneath valve disk. Operators placed plant in stable conditions using EOPs. W/980709 ltr.

DISTRIBUTION CODE: IE22T COPIES RECEIVED: LTR 1 ENCL 1 SIZE: 5
TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

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